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ATTN: Document Control Desk
U.S. Nuclear Regulatory Commission
Washington DC 20555-0001

Subject: Duke Energy Carolinas, LLC
Oconee Nuclear Station
Docket Numbers 50-269, 50-270, and 50-287
Technical Specification Bases Change 2017-04

The attached change to the Oconee Nuclear Station Technical Specification Bases was processed in accordance with the provisions of Technical Specification 5.5.15, "Technical Specifications (TS) Bases Control Program."

Technical Specification Bases change 2017-04 adds the NRC-approved COPERNIC fuel performance code, which directly accounts for thermal conductivity degradation with burnup, into the licensing basis. Also, Safety Limit 2.1.1.1 is modified by a NOTE stating that, following transition to the COPERNIC Fuel Performance Methodology, both the TACO and GDTACO Fuel Performance Methodologies will no longer be applicable.

Any questions regarding this information should be directed to Stephen Newman, Oconee Regulatory Affairs, at (864) 873-4388.

Sincerely,

Thomas D. Ray

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Vice President
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Attachments

ADD1
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Attachments

TSB List of Effective Pages (LOEPs), Rev. 015		LOEP	1 - 4
TSB 2.1.1	Reactor Core SLs Rev. 001	2.1.1	1 - 4

SECTION/PAGES	REVISION NUMBER	IMPLEMENTATION DATE
TOC	000	09/03/14
B 2.1.1	001	06/08/17
B 2.1.2	000	02/06/14
B 3.0	000	10/20/11
B 3.1.1	000	05/16/12
B 3.1.2	000	05/16/12
B 3.1.3	000	06/02/99
B 3.1.4	000	07/23/12
B 3.1.5	000	05/16/12
B 3.1.6	000	07/23/12
B 3.1.7	000	07/23/12
B 3.1.8	000	05/16/12
B 3.2.1	000	05/16/12
B 3.2.2	000	05/16/12
B 3.2.3	000	05/16/12
B 3.3.1	003	01/17/17
B 3.3.2	000	12/14/04
B 3.3.3	000	12/10/14
B 3.3.4	000	12/10/14
B 3.3.5	000	12/10/14
B 3.3.6	000	12/10/14
B 3.3.7	000	12/10/14
B 3.3.8	000	05/16/12
B 3.3.9	000	05/16/12
B 3.3.10	000	05/16/12
B 3.3.11	001	01/17/17
B 3.3.12	000	05/16/12

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B 3.3.13	000	05/16/12
B 3.3.14	001	01/17/17
B 3.3.15	000	05/16/12
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B 3.3.21	000	05/16/12
B 3.3.22	000	05/16/12
B 3.3.23	000	05/16/12
B 3.3.24	000	09/26/01
B 3.3.25	000	11/05/03
B 3.3.26	000	11/05/03
B 3.3.27	000	12/10/14
B 3.3.28	000	05/16/12
B 3.4.1	000	05/16/12
B 3.4.2	000	12/16/98
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B 3.4.6	001	04/18/17
B 3.4.7	001	04/18/17
B 3.4.8	001	04/18/17
B 3.4.9	000	05/16/12
B 3.4.10	001	09/21/15
B 3.4.11	000	10/12/12
B 3.4.12	000	06/13/14

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B 3.4.13	001	01/17/17
B 3.4.14	001	09/21/15
B 3.4.15	001	11/24/15
B 3.4.16	001	08/23/16
B 3.5.1	000	05/16/12
B 3.5.2	003	04/18/17
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B 3.5.4	000	05/16/12
B 3.6.1	001	01/17/17
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B 3.7.9	000	08/28/14
B 3.7.10	003	01/17/17
B 3.7.10a	001	01/17/17
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B 3.7.12	000	05/16/12
B 3.7.13	000	08/19/10
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B 3.7.15	000	10/24/07
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B 3.7.17	001	01/17/17
B 3.7.18	000	06/15/06
B 3.7.19	001	03/10/16
B 3.8.1	002	01/17/17
B 3.8.2	000	04/07/11
B 3.8.3	001	01/17/17
B 3.8.4	000	12/18/07
B 3.8.5	000	05/16/12
B 3.8.6	000	05/16/12
B 3.8.7	000	05/16/12
B 3.8.8	001	01/17/17
B 3.8.9	001	01/17/17
B 3.9.1	000	05/16/12
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B 3.9.3	001	01/17/17
B 3.9.4	002	04/18/17
B 3.9.5	001	04/18/17
B 3.9.6	000	05/16/12
B 3.9.7	000	05/16/12
B 3.9.8	000	06/25/14
B 3.10.1	001	01/17/17
B 3.10.2	000	11/05/14

Note: With the introduction of Fusion in June 2015, all controlled documents require a three-digit revision number. Thus, the revision numbers were set to "000" in the summer of 2015. As such, the revision dates for Revision 000 are based on the implementation dates for revisions in effect prior to this change.

B 2.0 SAFETY LIMITS (SLs)

B 2.1.1 Reactor Core SLs

BASES

BACKGROUND

ONS Design Criteria (Ref. 1) require that reactor core SLs ensure specified acceptable fuel design limits are not exceeded during steady state operation, normal operational transients, and anticipated transients. This is accomplished by having a departure from nucleate boiling (DNB) design basis, which corresponds to a 95% probability at a 95% confidence level (95/95 DNB criterion) that DNB will not occur and by requiring that the fuel centerline temperature stays below the melting temperature.

DNB is not a directly measurable parameter during operation, but neutron power and Reactor Coolant System (RCS) temperature, flow and pressure can be related to DNB using a critical heat flux (CHF) correlation. The BWC (Ref. 2), the BWU (Ref. 4), and the BHTP (Ref. 5) CHF correlations have been developed to predict DNB for axially uniform and non-uniform heat flux distributions. The BWC correlation applies to Mark-BZ fuel. The BWU correlation applies to the Mark-B11 fuel. The BHTP correlation applies to the MARK-B-HTP fuel. The local DNB heat flux ratio (DNBR), defined as the ratio of the heat flux that would cause DNB at a particular core location to the actual local heat flux, is indicative of the margin to DNB. The minimum value of the DNBR, during steady-state operation, normal operational transients, and anticipated transients is limited to 1.18 (BWC), 1.19 (BWU) and 1.132 (BHTP).

The restrictions of this SL prevent overheating of the fuel and cladding and possible cladding perforation that would result in the release of fission products to the reactor coolant. Overheating of the fuel is prevented by maintaining the steady state peak linear heat rate (LHR) below the level at which fuel centerline melting occurs. Overheating of the fuel cladding is prevented by restricting fuel operation to within the nucleate boiling regime, where the heat transfer coefficient is large and the cladding surface temperature is slightly above the coolant saturation temperature.

Fuel centerline melting occurs when the local LHR, or power peaking, in a region of the fuel is high enough to cause the fuel centerline temperature to reach the melting point of the fuel. Expansion of the pellet upon centerline melting may cause the pellet to stress the cladding to the point of failure, allowing an uncontrolled release of activity to the reactor coolant. The maximum allowable fuel centerline temperatures are given by the relationships defined in Reactor Core SL 2.1.1.1 and are dependent on whether TACO3, GDTACO, or COPERNIC analyses are

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BACKGROUND (continued)

utilized. For TACO3 applications, the dependency of the fuel melt temperature on the as-built oxygen-to-uranium ratio for UO_2 fuel pins is provided by the fuel vendor. For gadolinia fuel pins, there is no dependence on the oxygen-to-uranium ratio.

Operation above the boundary of the nucleate boiling regime could result in excessive cladding temperature because of the onset of DNB and the resultant sharp reduction in heat transfer coefficient. Inside the steam film, high cladding temperatures are reached, and a cladding-water (zirconium-water) reaction may take place. This chemical reaction results in oxidation of the fuel cladding to a structurally weaker form. This weaker form may lose its integrity, resulting in an uncontrolled release of activity to the reactor coolant.

The proper functioning of the Reactor Protection System (RPS) and main steam relief valves (MSRVs) prevents violation of the reactor core SLs.

APPLICABLE SAFETY ANALYSES

The fuel cladding must not sustain damage as a result of normal operation and anticipated transients. The reactor core SLs are established to preclude violation of the following fuel design criteria:

a. There must be at least 95% probability at a 95% confidence level (95/95 DNB criterion) that the hot fuel rod in the core does not experience DNB; and

b. The hot fuel pellet in the core must not experience fuel centerline melting.

The RPS setpoints (Ref. 3), in combination with all the LCOs, are designed to prevent any analyzed combination of transient conditions for RCS temperature, flow and pressure, and THERMAL POWER level that would result in a DNB ratio (DNBR) of less than the DNBR limit and preclude the existence of flow instabilities.

Automatic enforcement of these reactor core SLs is provided by the following:

a. RCS High Pressure trip;

b. RCS Low Pressure trip;

c. Nuclear Overpower trip;

d. RCS Variable Low Pressure trip;

e. Reactor Coolant Pump to Power trip;

BASES

APPLICABLE SAFETY ANALYSIS (continued)

f. Flux/Flow Imbalance trip;

g. High Core Outlet Temperature trip; and

h. MSRVs.

The SL represents a design requirement for establishing the RPS trip setpoints identified previously.

SAFETY LIMITS

SL 2.1.1.1 and SL 2.1.1.2 ensure that fuel centerline temperature stays below the melting point and that the minimum DNBR is not less than the safety analyses limit. Safety Limit 2.1.1.1 is modified by a NOTE indicating that following transition to the COPERNIC Fuel Performance Methodology both the TACO and GDTACO Fuel Performance Methodologies are not applicable.

The SLs are preserved by monitoring process variables, AXIAL POWER IMBALANCE and Variable Low RCS Pressure, to ensure that the core operates within the fuel design criteria. AXIAL POWER IMBALANCE and Variable Low RCS Pressure protective limits are provided in the COLR. The trip setpoints are derived by adjusting the measurement system independent AXIAL POWER IMBALANCE and Variable Low RCS Pressure protective limits given in the COLR to allow for measurement system observability and instrumentation errors.

Operation within these limits is ensured by compliance with the AXIAL POWER IMBALANCE and Variable Low RCS Pressure protective limits preserved by their corresponding RPS setpoints in LCO 3.3.1, "Reactor Protection System (RPS) Instrumentation," as specified in the COLR. The AXIAL POWER IMBALANCE protective limits are separate and distinct from the AXIAL POWER IMBALANCE operating limits defined by LCO 3.2.2, "AXIAL POWER IMBALANCE Operating Limits." The AXIAL POWER IMBALANCE operating limits in LCO 3.2.2, also specified in the COLR, preserve initial conditions of the safety analyses but are not reactor core SLs.

APPLICABILITY

SL 2.1.1.1 and SL 2.1.1.2 only apply in MODES 1 and 2 because these are the only MODES in which the reactor is critical. Automatic protection functions are required to be OPERABLE during MODES 1 and 2 to ensure operation within the reactor core SLs. The MSRVs, or automatic protection actions, serve to prevent RCS heatup to reactor core SL conditions or to initiate a reactor trip function, which forces the unit into MODE 3. Setpoints for the reactor trip functions are specified in LCO 3.3.1.
In MODES 3, 4, 5, and 6, Applicability is not required, since the reactor is not generating significant THERMAL POWER.

BASES (continued)

SAFETY LIMIT VIOLATIONS

The following SL violation responses are applicable to the reactor core SLs.

<u>2.2.1</u>

If SL 2.1.1.1 or SL 2.1.1.2 is violated, the requirement to go to MODE 3 places the unit in a MODE in which these SLs are not applicable.

The allowed Completion Time of 1 hour recognizes the importance of bringing the unit to a MODE of operation where these SLs are not applicable and reduces the probability of fuel damage.

REFERENCES

1. UFSAR, Section 3.1.
2. BAW-10143P-A, "BWC Correlation of Critical Heat Flux," April 1995.
3. UFSAR, Chapter 15.
4. BAW-10199P, "The BWU Critical Heat Flux Correlations," Addendum 1, April 2000
5. BAW-10241(P)(A), Revision 1, BHTP DNB Correlation Applied with LYNXT, Framatome ANP, July 2005.